NON-PUBLIC?: N
ACCESSION #: 9512190383
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Big Rock Point Plant PAGE: 1 OF 04

DOCKET NUMBER: 05000155

TITLE: AUTOMATIC REACTOR SCRAM DURING PLANT STARTUP
EVENT DATE: 11/16/95 LER #: 95-007-00 REPORT DATE: 12/14/95

OTHER FACILITIES INVOLVED: N/A DOCKET NO: 05000

OPERATING MODE: N POWER LEVEL: 022

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Michael D Bourassa, Licensing TELEPHONE: (616) 547-8244
Supervisor

COMPONENT FAILURE DESCRIPTION:
CAUSE: A SYSTEM: JE COMPONENT: MANUFACTURER:
REPORTABLE NPRDS: N

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On November 16, 1995, the facility was being returned to service following a short maintenance outage taken to repair leaking main condenser tubes. With the reactor power at approximately 22%, an automatic reactor scram occurred at 0315. The scram was automatically initiated (by the reactor protection system) due to a high rate of power change caused by excessive feedwater flow shortly after the controller had been switched from manual to automatic operation. The operators completed all scram actions, and established a primary system cooldown rate less than the 100 degrees F per hour required by Technical Specifications.

The root cause of this event was failure to place the feedwater regulating valve in automatic using a bumpless transfer. Performance of multiple activities during plant startup contributed to the unsuccessful feedwater control transfer. Prior to restarting the facility,

Operations' procedures O-TGS-1, Master Checklist, and Standard Operating Procedure 16, Feedwater System were revised to offer more guidance during this segment of plant startup.

The facility was returned to service on November 17, 1995.

END OF ABSTRACT

IDENTIFICATION OF EVENT

This event is reportable to the Nuclear Regulatory Commission pursuant to:

1) 10 CFR 50.72(b)(2)(ii) - Any event or condition that results in a manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection Sy
tem (RPS).

2) 10 CFR 50.73(a)(2)(iv) - Any event or condition that resulted in a manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS).

DESCRIPTION OF EVENT

On November 16, 1995, the facility was being returned to service following a short maintenance outage taken to repair leaking main condenser COND! tubes. Three significant activities were being conducted in the control room NA!. The turbine/generator TG! was about to be synchronized to the power grid, power was being increased by withdrawing control rods AA!, and the feedwater regulating valve SJ;FCV! was being changed from manual control closed position, to automatic control open position. Plant conditions at 0300 were as follows:

- Reactor RCT! power was approximately 50 megawatt thermal, or 22% of rated power.

- Steam flow through the turbine bypass valve TRB;FCV! was approximately 160,000 lbm/hr.

- Feedwater SJ! flow was approximately 90,000 lbm/hr.

- Steam Drum SD! level was plus one inch over centerline, dropping one inch every two minutes.

- Turbine was rotating at its rated 3600 rpm.

- The number 2 feedwater SJ;P! pump was operating.

- The feedwater regulating valve was in manual and closed; and

- The feedwater regulating bypass valve was closed.

At 0309 the feedwater regulating valve was placed to the auto position with the steam drum level at plus one inch above the centerline. The steam drum level decreased to approximately minus 2 inches below the centerline when the feedwater regulating valve started to receive an open signal from the controller to respond to the decreasing steam drum level and the low feedwater flow; however it did not open as expected by the operating crew. Before the control room operators could return the feedwater regulating valve to manual as directed by the shift supervisor, the feedwater regulating valve opened at a rapid rate, causing a cold water addition to the core and the start of the standby condensate pump SD;P!. Reactor power increased from approximately 50 megawatt thermal up to 120 megawatts thermal, at a rate greater than the reactor trip parameter of 50 megawatts thermal per minute as monitored by the wide range reactor power monitors JD!. This condition resulted in an automatic reactor scram at 0315. The operators completed all scram actions, and established a primary system cooldown rate below 100 degrees F per hour as required by Technical Specifications.

ROOT CAUSE

The root cause of this event was failure to place the feedwater regulating valve in automatic using a bumpless transfer. Performance of multiple activities during plant

startup contributed to the unsuccessful feedwater control transfer. The methods utilized to evaluate this event included interviews with plant staff on duty at the time of the event, a human factors analysis, and an engineering analysis.

Human factors analysis

Shortly following the event, the operating crew was interviewed and the following conclusions were reached:

- There was no evidence of management or self-induced pressure at the time of the event.

- Fatigue/illness was not a factor.

- The task was within the expected experience level of the operating crew.

- The overall condition of the plant and the activities surrounding the startup were understood by the control room crew.

- The coordination of placing the feedwater controller to automatic, withdrawing control rods, and adjusting the turbine bypass valve could have been performed separately instead of creating overlap.

- There could have been more discussion of what the consequences of the steep drop in steam drum level would have been if the feedwater regulating valve did not open.

- The Shift Supervisor was not the normal supervisor for the crew.

- There was evidence of change from past practices. Normally the feedwater regulating bypass valve would be open and the difference between feedwater and steam flow would be smaller.

Engineering/Maintenance analysis

The transient was well within plant design limits and no equipment or fuel damage was neither noted nor expected. An investigation of the feedwater regulating valve and its control system did not detect any abnormalities. However, since the facility restart, the feedwater regulating valve is exhibiting some operating inconsistencies that will require further investigation for root cause implications.

Other factors that contributed to the event were:

1. Lack of practice prior to performing these tasks both individually and as a team;

2. Lack of specific procedural guidance for performing the tasks.

3. Lack of knowledge of the expected responses and operating limits of the feedwater regulating valve.

4. Inadequate communication between team members performing tasks that dramatically affected each other.

5. Lack of contingency planning.

6. Potentially degraded feedwater regulating valve.

CORRECTIVE ACTION

Immediate

The operators completed all scram actions, and established a primary system cooldown rate less than 100 degrees F as required by Technical Specifications.

Prior to returning the unit to service, Operations' procedures O-TGS-1, Master Checklist, and Standard Operating Procedure 16, Feedwater System were revised to offer more guidance during this plant startup evolution. The plant was restarted November 17, 1995.

To Prevent Recurrence

1. Develop methods that can be used to practice infrequently performed tasks prior to performing them; for plant shutdowns and startups use the simulator or control room to walk through the process and procedures prior to performing the task(s) in the control room.

THIS ACTION WILL BE COMPLETE BY JANUARY 12, 1996

2. Implement the methods developed to practice infrequently performed task(s) prior to performing them in the control room.

THIS ACTION WILL BE COMPLETE BY MARCH 13, 1996.

3. Revise procedures 0-TGS-1 and SOP-16 to address the issues.

THIS ACTION WILL BE COMPLETED BY JANUARY 12, 1996.

4. Provide management expectations of the individuals' responsibility and the crew teamwork responsibility to keep all members of the team (crew and management) abreast of changing plant parameters along with appropriate management oversight. Also provide management expectations of the need to consider contingencies when operating plant equipment.

THIS ACTION WILL BE COMPLETED BY FEBRUARY 12, 1996.

5a. Review and evaluate the present training material (Classroom, simulator, OJT) for the control of feedwater flow and the feedwater regulating valve when starting up and shutting down to determine what corrective actions or revisions need to be made to the training materials. Determine if other components or operator actions need to be evaluated as a result of this incident.

5b. Train the operating crews on the expected responses and operating limits of the feedwater regulating valve and the past practice of putting it into service.

THESE ACTIONS WILL BE COMPLETED PRIOR TO STARTUP FROM THE 1996 REFUELING OUTAGE

6. Re-evaluate feedwater regulating valve and controller performance.

THIS ACTION WILL BE COMPLETED BY THE END OF THE 1996 REFUELING OUTAGE.

SAFETY SIGNIFICANCE

This event is bounded by the Updated Final Hazards Safety Report Section 15.1.2, Increase in Feedwater Flow, which assumes the reactor initial conditions are 102% power and 1350 psia. The feedwater flow rate just prior to the reactor scram was less than 10% of full power flow. Reactor power was approximately 20% of design and the primary system pressure was about 800 psia. Core physics package calculations were also performed to determine if the fuel could have been affected. Preconditioning power ramp rates are only required for power in excess of 165 megawatts thermal. Since the power remained below 165 megawatts thermal, the high power ramp rate had no affect on the fuel.

ATTACHMENT TO 9512190383 PAGE 1 OF 1

Consumers
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Big Rock Point Nuclear Plant, 10269 US-31 North, Charlevoix, MI 49720

December 14, 1995

Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

DOCKET 50-155 - LICENSE DPR-6 - BIG ROCK POINT PLANT - LICENSEE EVENT
REPORT 95-007: AUTOMATIC REACTOR SCRAM DURING PLANT STARTUP.

LICENSEE EVENT REPORT 95-007, AUTOMATIC REACTOR SCRAM DURING PLANT
STARTUP, is attached. This event is reportable to the Nuclear Regulatory Commission pursuant to:

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Gregory C Withrow
Director Plant Safety & Licensing

CC: Administrator, Region III, USNRC
NRC Resident Inspector - Big Rock Point

ATTACHMENT

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